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Application of MCNP code to shielding analysis for single and multilayers of polyethylene and lead

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Abstract : Calculations were made using MCNP-4B code to study the neutron shielding effectiveness of polyethylene (PE) and lead (Pb) as a single and multilayer arrangements against U^{235} fission neutron source. Shielding slabs of 5 cm thickness were employed successively Transmission factors for the above-mentioned shielding setups were determined upto 100 cm thickness. Dose spectra have been calculated for different thicknesses of the shielding arrangements. The results revealed that using slabs successively of 15 cm PE plus 15 cm Pb is as efficient in shielding fission neutrons as using of 30 cm PE shield alone It was also found that after 40 cm shield thickness, the removal cross section values of the multilayers and single PE shield are not very much different, and are approaching each other with increasing shield thickness.

Keywords : MCNP, polyethylene, lead, neutron shielding

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Polyethylene (PE) and lead (Pb) are well-known materials essential for neutron and gamma ray shielding. Several groups and workers compiled tables, graphs and formula to guide in the design of radiation protection system for neutron sources [1,2]. The normal practice in neutron shielding is to decrease the source neutron energy to such a value that they are easily absorbed by nuclei present in the shield. But it is observed that high-energy gamma rays (up to 10 MeV or more) are produced as a result of neutron absorption reactions. In addition, high-energy neutrons (in the MeV range) mostly lose their energy through inelastic scattering reaction that also produces gammas. Therefore, it is always essential in neutron shielding work to use materials that will simultaneously attenuate both neutron and gamma radiations.

An optimum combination of PE and Pb is of great importance to reduce biological dose rates to predetermined levels [3]. Hence, different shielding arrangements of single and multilayers of PE and Pb have been analyzed to obtain various shielding parameters; mainly, transmission factor, removal cross section and dose rate. The analysis to find the above mentioned parameters have been performed using the MCNP-4B computer transport Code.

MCNP-4B is a three-dimensional, point wise, continuous energy cross section Monte Carlo Code, which is capable of performing neutron, photon, or coupled neutron/photon transport calculations [4]. The code uses continuous-energy, nuclear and atomic data libraries, the primary sources of which are evaluations from the Evaluated Nuclear Data File (ENDF) [5,6], the Activation Library (ACTL) [7] compilations from Livermore, and evaluations from the Applied Nuclear Science Group (ANS Group) [8-10] at Los Alamos. The evaluated data are processed into a format appropriate for MCNP with the help of codes such as NJOY [11,12]. Nuclear data tables exist for different types of neutron interactions from which appropriate ones may be selected through unique identifiers for each table, called ZAIDs.

Continuous nuclear cross section data based on the ENDF/B-V were used for the computations, including the appropriate thermal neutron scattering functions $S(\alpha,\beta)$. A Pentium-3 PC with 600 MHz clock speed and 512 Ram was used for all the

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calculations. Lahey Fortran Complier and 386 Link program were used to make this code operational in Pentium-3.

The neutron fluxes were computed for different shielding thickness of lead and polyethylene. These flux values then used to evaluate transmission factor and removal cross section for each shielding arrangement. Dose rate in μSv/h was also calculated by using flux to dose conversion factor taken from Oak Ridge MCNP Manual [13,14]. Calculations were performed using a fixed U^{235} point source with a maximum of 2 million histories to have the statistical error less than 4% for all the cases. The distance of the point source from the target used in the calculation for all the shielding setup cases was about 1 mm meaning that we placed the shielding slab just after the source.

The calculation of the neutron streaming flux through any shielding material by using the MCNP Code involves simulation of the MCNP geometry model, reflecting fully the real physical situation. A three-dimensional geometry model was employed for this MCNP calculation. The composition of air was taken to be oxygen (20.95% by weight) and nitrogen (78.08% by weight) only [15]. Considering 100% abundance the density of Pb was taken to be 11.34 gcm^{-3}. The density of PE was taken to be 0.95 gcm^{-3} containing 85.7% carbon and 14.3% hydrogen by weight [16].

The geometry is symmetric with respect to the X-Z center plane and the Y-Z center plane. The X and Z directions were taken to be unaltered while the shielding slabs were sequentially increased along the plane perpendicular to the y-direction to obtain the corresponding neutron fluxes. The lateral dimensions of the geometry were 100 cm x 100 cm.

The transmission characteristics of the fission neutrons through polyethylene and lead for both single and multilayer arrangements have been investigated. The characteristics of the transmission factors of those arrangements are depicted in Figure 1.

The behavior of fission neutrons for the multilayers (PE + Pb) and (Pb + PE) lie between its composites (PE and Pb) up to a thickness of about 45 cm beyond which the multilayer curves practically coincide with that of the (PE). From Figure 1, it appears that at 15 cm thickness the multilayer (PE + Pb) transmits (24.2 %) and the multiplayer (Pb + PE) transmits (37.0 %) while their composites (PE) and (Pb) transmits (13.2 %) and (49.1 %) respectively. At 25 cm shield thickness, again the multiplayer (PE + Pb) transmits only (6.9 %) and (Pb + PE) transmits (12.3 %) as against (2.1 %) and (3.9 %) for PE and (Pb) respectively. It is also observed from Figure 1 that when equal thickness of PE and Pb were combined in the multilayers, then almost similar value of the transmission factors were obtained, *e.g.* at 10, 20, 30 and 40 cm shield thicknesses.

In the investigation of thickness-dependence of removal cross sections of PE and Pb, and their multilayers, it is observed that for the PE single shield, there was a sharp increase up to a thickness of 20 cm and thereafter the rate of increment remained almost constant. However, in the case of single Pb shield almost an opposite feature was observed. The cross section for the multilayers of (PE + Pb) and (Pb + PE) are almost similar. The values are shield thickness dependent taking a zigzag pattern. The higher values correspond to PE while the lower ones correspond to Pb.

Dose spectra have been calculated from the corresponding flux spectra at different shield thicknesses. The fluxes were converted into dose rate spectra by multiplying the neutron flux values of different energies by the corresponding dose factor values [13]. Two typical dose spectra are shown in Figures 2 and 3.

The common feature observed for all multilayer shields is that, they show two peaks one at 2 MeV and the other at 5 MeV, and an absorption trough at 2.5 MeV. The relative magnitudes

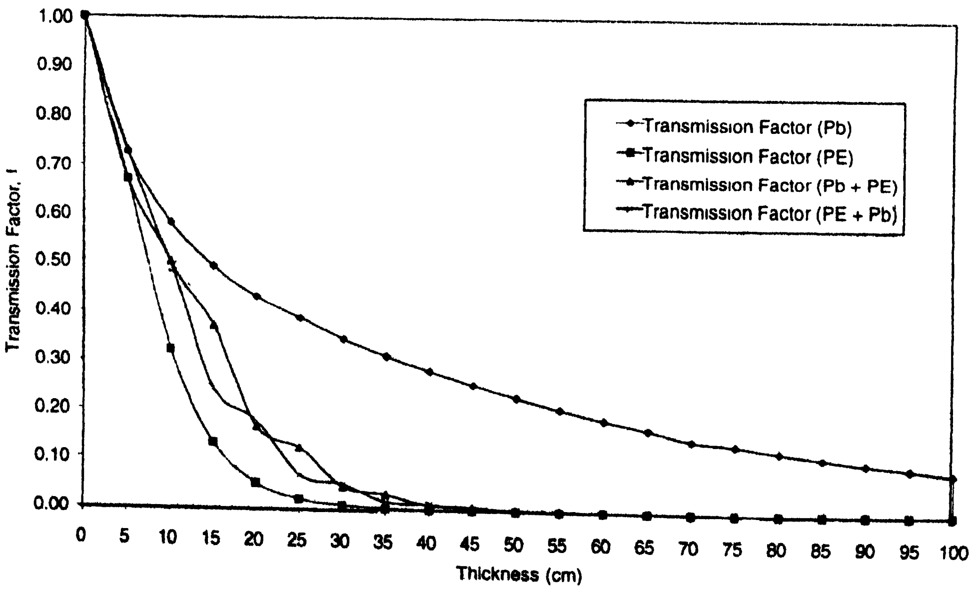

Figure 1. Variation of transmission factor with thickness for PE, Pb, Pb + PE, and PE + Pb.

of the peaks decrease with the increase of shield thickness producing long tail between 7 and 14 MeV. The first peak appears at 2 MeV only for 5 and 10 cm shield thickness, and for multilayer shield thickness it shifts towards lower energy. It may be due to the accumulation of thermal and epithermal neutrons. Pb is used with 5 cm PE compared with that of the reverse combination.

Figure 3 shows the dose rate spectra of single and multilayer shielding arrangements at 10 cm shielding thickness.

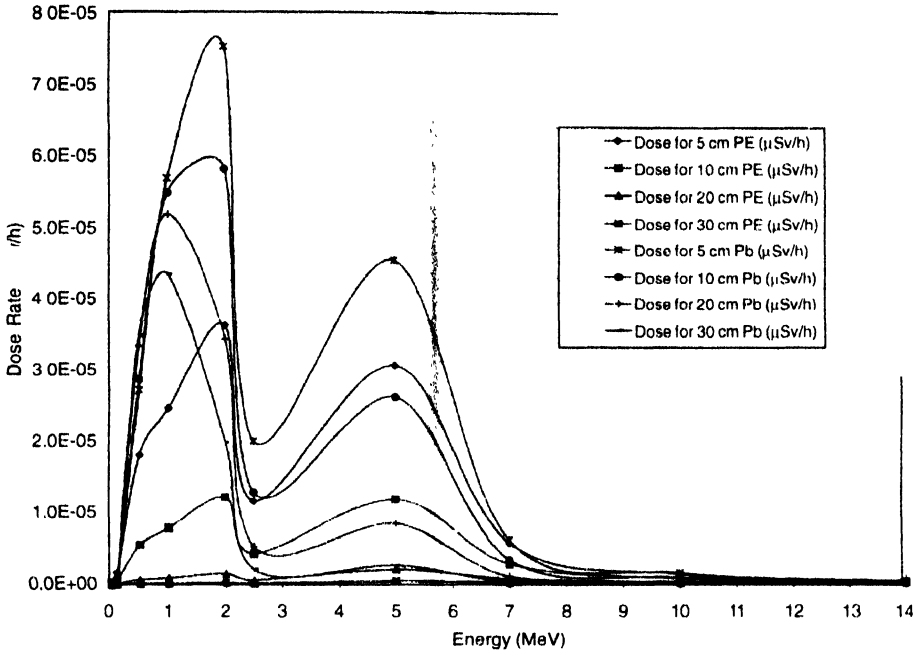

Figure 2. Dose spectra of fission neutrons for the shields PE and Pb at (5, 10, 20, and 30) cm

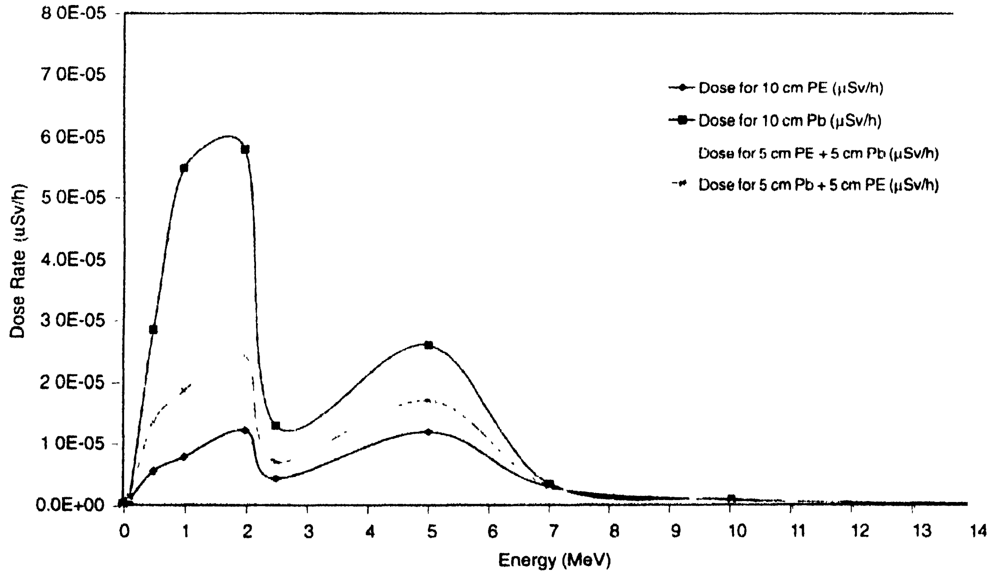

Figure 3. Dose spectra of fission neutrons for the shields PE , Pb, PE + Pb, and Pb + PE at 10 cm

In Figure 2, a comparison of dose rate for PE and Pb with shield thickness is shown. The same common feature was also observed for single shields. It is further observed that when components of equal thickness is used for building multilayer shield the dose rate remains unaltered irrespective of the order of single shield component. It is noticed that in the energy range 0 - 2 MeV, the dose rate becomes more than double when 10 cm

In this figure, two things are worth mentioning. Firstly, dose rate values for Pb do not decrease appreciably with thickness. Secondly, the dose rate values for Pb is much larger than the other combination while those for PE is the lowest and the multilayers' dose rate values remain between the single shields with its values nearer to the PE dose rate values.

The following observations can be made from our study: dose rate for Pb does not decrease appreciably with increasing thickness and it is much larger than the other combination; for PE it is the lowest; after 8 MeV the dose rate values are negligibly small and almost the same for all shield setups.

The analytically calculated results reveal that using succeeding slabs of 5 cm thickness of polyethylene and lead in either order would be as efficient as single polyethylene for shielding of fission neutrons. In particular, after about 30 cm thickness, the multilayer and PE shield set-ups give almost the same shielding effectiveness. But the advantage of using lead as component of the multilayer will simultaneously attenuate gamma rays very effectively. Therefore, total dose rate reduction will be best controlled using a succeeding PE and Pb shield slabs rather than a single PE ones. It is also found that the removal cross section is thickness and energy dependent. In addition, the removal cross section values for the multilayers are not far from that of the single polyethylene shield especially at deep penetration.

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